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A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

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Power Reactor and Nuclear Fuel Development Corporation

1. Experimental Fast Reactor

1.1 General Status

This report covers the activities of the experimental fast reactor JOYO from April 1985 to March 1986. After completion of the 7th duty cycle operation at the end of March 1985, special operation was carried out for the in-vessel performance test of the failed fuel detection and location system (FFDL) by irradiating slitted pins, natural circulation test from 30 MWt, and in-core measurement of coolant flow rate of each core subassembly during April 1985.

The 5th annual inspection was carried out from the end of April 1985 to the beginning of December 1985, including disassembly and inspection of the double rotating plugs. Works for modification of the plant were done on such system as the control panel of the reactor protection system in order to enable inspection of the scram circuit during reactor operation, exchange of the cooling towers for the auxiliary service system and ventillation system, liquid waste processing system, and replacement of one of the in-vessel sodium level meters to a wide range level meter. Installation of the special irradiation equipments was also done for the instrumented test assembly (INTA) and the upper plug rig (UPR).

The 8th duty cycle operation started on Dec.2, 1985, and the 9th duty cycle is to be completed at the end of March 1986 along with natural circulation test from 75MWt.

Besides the operation and maintenance works, followings were conducted during the period.

1. Development of operation and maintenance supporting systems using computers.

2. Data supply to Centralized Reliability Data Organization (CREDO) of U.S.A. based on the agreement between DOE and PNC.

3. Application of the license to extend the fuel maximum burnup from 50,000 MWd/t to 75,000 MWd/t, and an increase of uranium enrichment of the fuel from 12wt% to 18wt%.

Operation history of JOYO is illustrated in Fig 1.1

1.2 Demonstration Test of Gas Tagging Method

A demonstration test of the gas tagging method simulating fuel failure was carried out at JOYO in April 1985. In this test, a dummy fuel pin with a slit which contained a tag gas capsule was loaded in the core and a fraction of the tag gas was released to the reactor cover gas at the reactor power of 3MWt. The tag gas was collected and enriched by the cryogenic adsorption bed with activated charcoal.

In this method, after release of the tag gas from the pin, a large charcoal trap is opened to the cover gas system, and adsorbs the tag gas contained in the cover gas at -180℃. The gas is desorbed by warming the trap, and readsorbed with another small trap. Then, the gas is purified and desorbed by helium gas flushing at -80℃, and collected into a measurement vial. The schematic diagram of the tag gas enriching system is shown in Fig. 1.2.

By means of mass spectrometric analysis, it was confirmed that the measured contents of the collected tag gas agreed with those of as built within the required precision. (see Table 1.1) As the result, the method of cryogenic adsorption developed by PNC was proved to be for practical use, and some experimental data were obtained on behaviors of tag gas in the reactor vessel.

1.3 Performance Test of FFDL using Slitted Fuel Pins

An FFDL (Failed Fuel Detection and Location System) employing in-core wet sipping method has been developed in JOYO. The sipping and sniffing mechanism of the FFDL is illustrated in Fig. 1.3, and overview of the FFDL is shown in Fig. 1.4.

In order to verify the performance of the FFDL, an in-pile simulation test was carried out using two identically slitted fuel pins. The slit is 1.0mm in length and 0.1mm in width, and perforated on the fuel claddings at the gas plenum position. The reactor power history during the test is shown in Fig. 1.5. Fig. 1.6 shows the count rate ratio of ^{113}Xe γ-ray from the test assembly by operation of the FFDL.

The major results of the test and the analyses are as follows.

(1) In the FFDL operation, a signal level of the test assembly was several hundreds times of the background which was measured for other core subassemblies. Thus, it was confirmed for the FFDL to have a capability to identify the failed fuel with defect at gas plenum.

(2) Measured amount of ^{133}Xe released from the test pins to the cover gas by the FFDL operation was 50% of the predicted with an assumption of the

FP release-to-birth ratio to be 0.05. Consequently, the sipping mechanism of the FFDL from the defect at gas plenum was proved.

(3) Measured amount of ^{133}Xe introduced into the gas circulation unit of the FFDL was 10% of the predicted. This low sampling efficiency may be due to the deat time of 7 seconds between release of the sipping port from the assembly and sodium introduction (see Fig. 1.3). Most of the FP gas in the test pins might be released during this dead time.

Further tests of the FFDL for pins with slit at fuel column are planned to be carried out in the near future.

1.4 Natural Circulation Test

A series of natural circulation tests in JOYO MK-II core were planned to be carried out from the 5th duty cycle in order to verify the capability of decay heat removal by natural circulation, and to obtain information of thermal hydraulic behavior in the reactor vessel. These tests are to be conducted stepwise from low to high reactor power with the combination of steady tests (constant reactor power) and transient tests (reactor trip), as shown in Table 1.2. Test II-B, the second step of the preliminary tests, was carried out from 30MW reactor power with trip of all main pumps in April 1985.

The major results of this test and analysis were as follows;

(1) Peak temperature of the central subassembly outlet coolant during the test was 448 ℃ (at 190 seconds after the pump trip), which was almost the same as that of MK-I core (Test D). The prediction by 2-dimensional analysis used COMMIX-1A code were 444 ℃ (at 176 seconds).

(2) The measured temperatures of outlet coolant of the subassemblies (row 3 ~ 6) splitted into two or three groups in the respective core flow zone. The cause of this phenomenon was the effects of cross flow from adjacent subassembly and upper plenum, presumably.

(3) Flow rates of natural circulation were 1.3% and 2.2% of the rated flow in the loop A and loop B of the primary cooling system, respectively, and 3% in the secondary cooling system.

1.5 Measurement of In-Core Coolant Flow Distribution

Coolant flow distribution in the MK-II core was measured after the 7th duty cycle operation in April 1985. Purposes of the measurement were to reconfirm the measured results of the MK- II initial core, to obtain data concerning burnup dependences of pressure drops of fuel assemblies, and to supply data for the natural circulation tests in JOYO.

Coolant flow of each core subassembly was measured using an electromagnetic flow meter mounted on the rotating plug. Flow rates of the primary cooling system were changed from 90% to 0% (all pumps stopped).

The measured results were as follows. (see Fig. 1.7)

1. In case of the pump-stop condition, flow rates of fuel assemblies were propotional to their own decay heat.
2. Flow distribution in the core was flatter at very low flow rate condition, 6% of the rated, than those of other conditions.
3. The burnup dependence of the pressure drop of fuel assembly was very small, less than -0.1% in the flow rates per 10^4 MWd/t of the fuel assmblies.
4. The calculated resutls of the flow distribution agreed satisfactorily with the measured for the condition of total flow rate more than 20% of the rated.

1.6 On-line measurement of Fuel Behavior under Reactor Operation

From the 8th duty cycle, an irradiation test using the Instrumented Test Assembly (INTA) started to obtain such in-core informations as fuel pellet centerline temperature, fission gas pressure in fuel pin, coolant flow and temperatures in subassembly, and neutron flux with real time under reactor operation.

These in-core informations are very important to investigate fuel characteristics and core characteristics.

Schematic view of the INTA and its instrumentations are shown in Fig. 1.8 and Fig. 1.9 respectively. Fig. 1.10 shows arrangement of the sensors attached to the fuel pins in the assembly.

At the end of the 8th duty cycle, this irradiation test are successfully in progress. These results will be used to improve analysis codes for fuel performance and core characteristics.

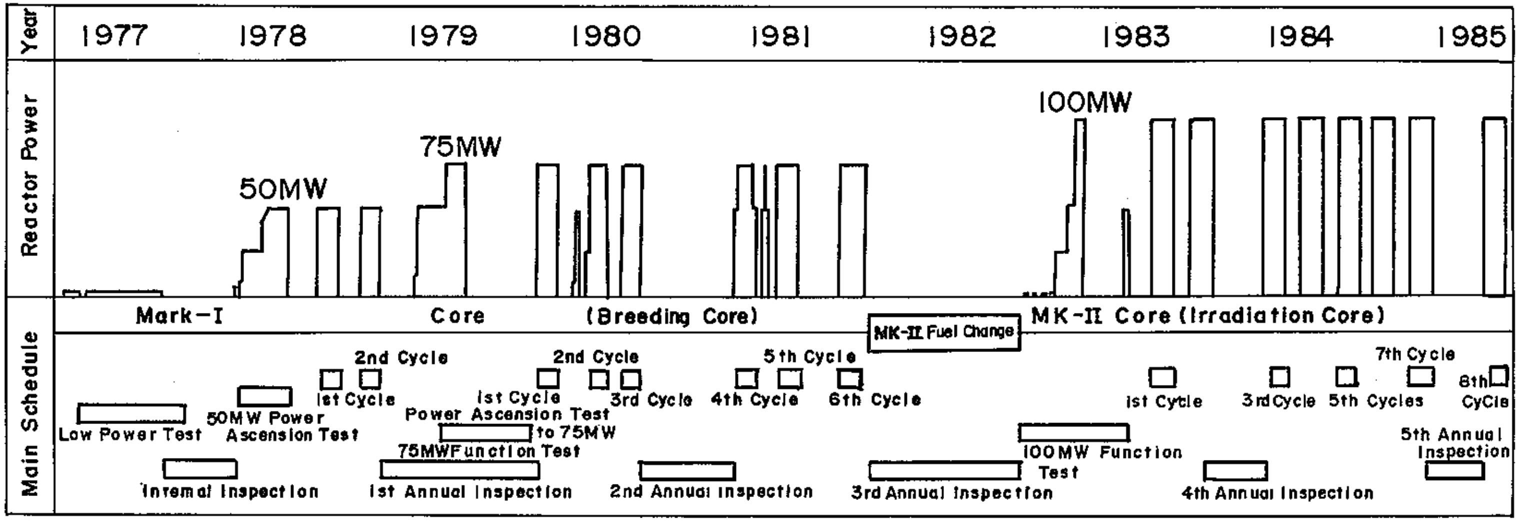


Fig. 1.1 Experimental Fast Reactor JOYO Operating History

Table 1.1 Analytical Results of Tag Gas

			as built	No. 1	No. 2
Kr	Isotopic Composition (%)	78	9.18	8.51	9.05
		80	34.40	34.13	33.95
		82	45.99	47.31	45.46
		83	8.11	7.53	8.15
		84	2.33	2.52	3.38
	Vol. (cc)		1.25	1.64×10^{-2}	9.12×10^{-3}
Xe	Isotopic Composition (%)	124	5.89	5.74	5.85
		126	2.72	2.98	2.67
		128	14.39	14.36	14.41
		129	68.89	67.84	68.79
		130	2.71	3.00	2.72
		131	3.71	4.31	4.01
		132	1.58	1.67	1.50
		134	0.11	0.11	0.04
	Vol. (cc)		1.25	1.66×10^{-2}	9.85×10^{-3}

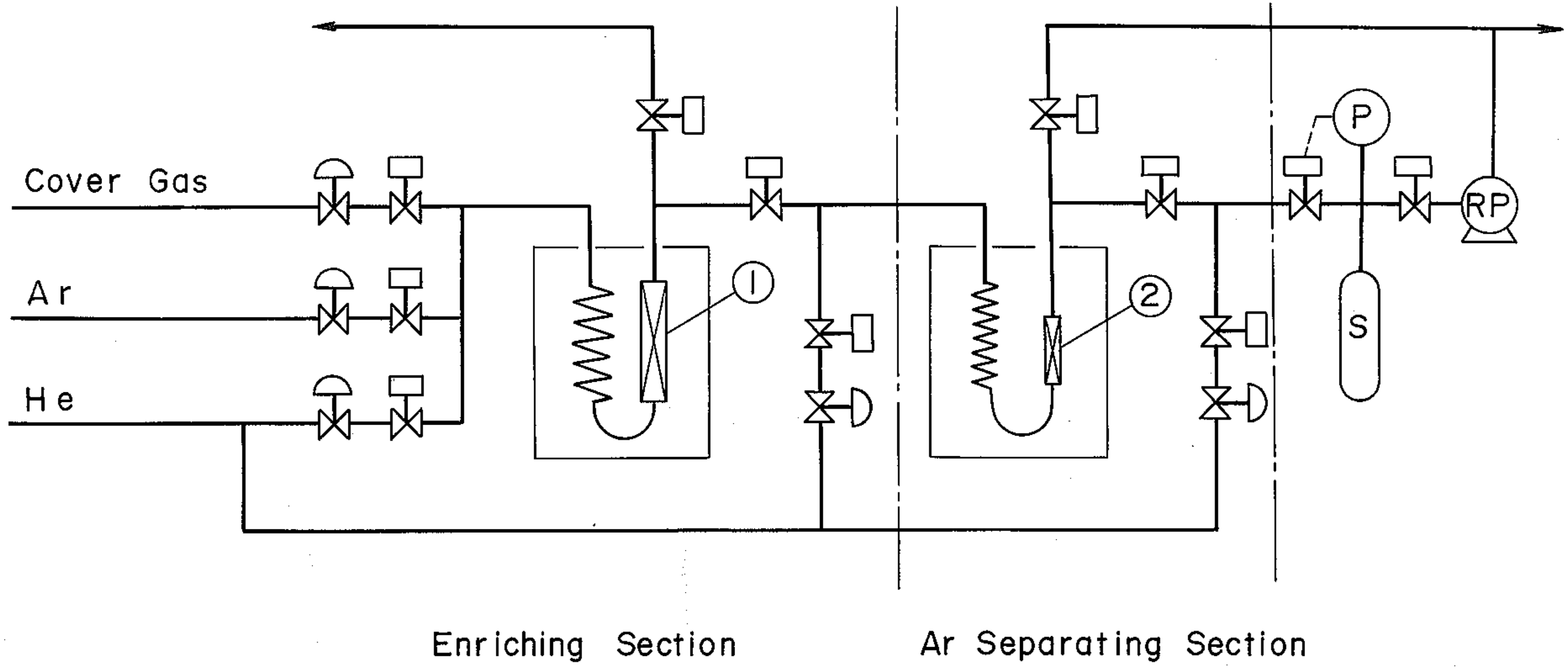


Fig. 1.2 Schematic Diagram of Tag Gas Enriching System

	Solenoid Valve
	Needle Valve
RP	Vacuum Pump
S	Measurement Vial
P	Pressure Gauge

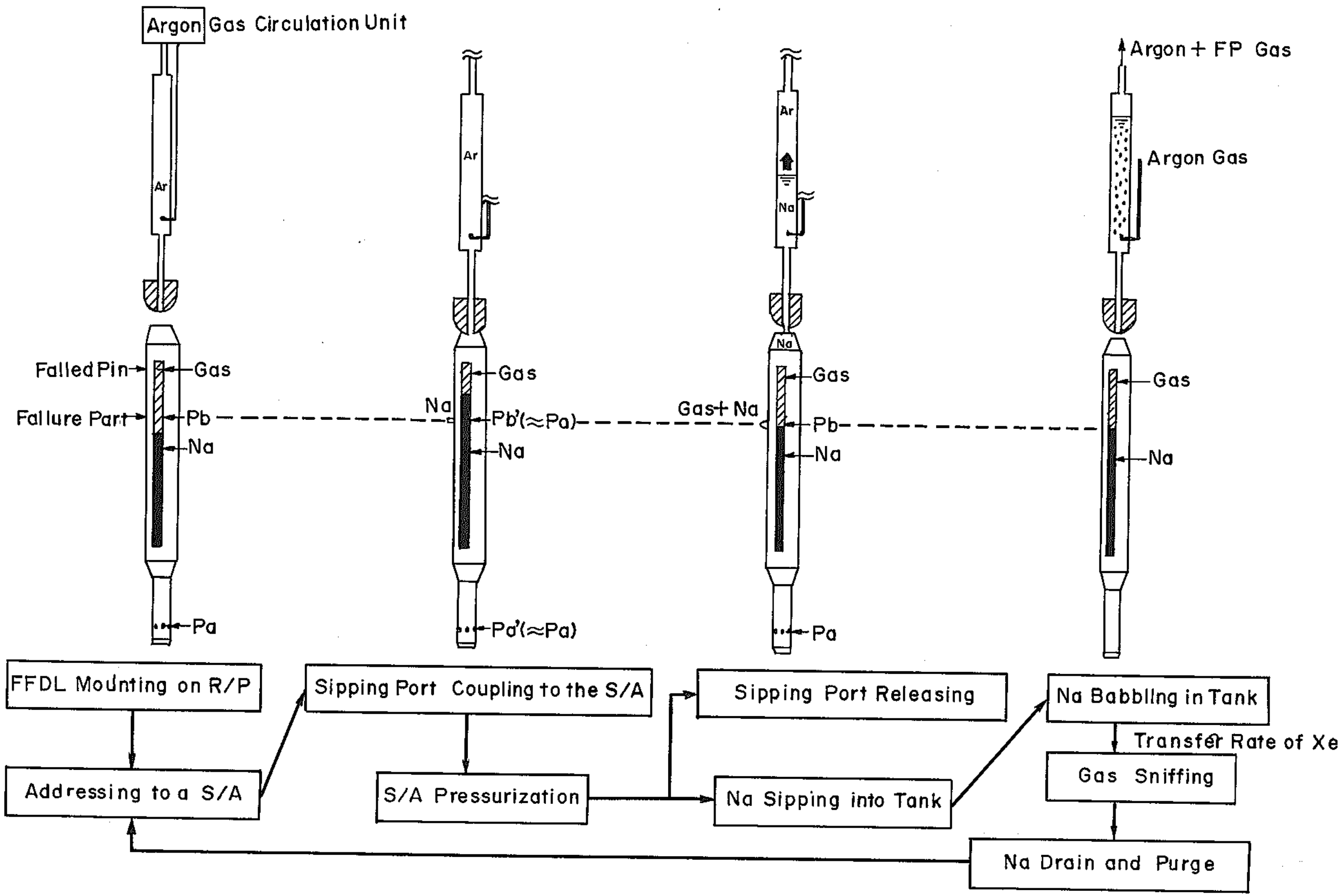


Fig.I.3 Sipping and Sniffing Mechanism of FFDL

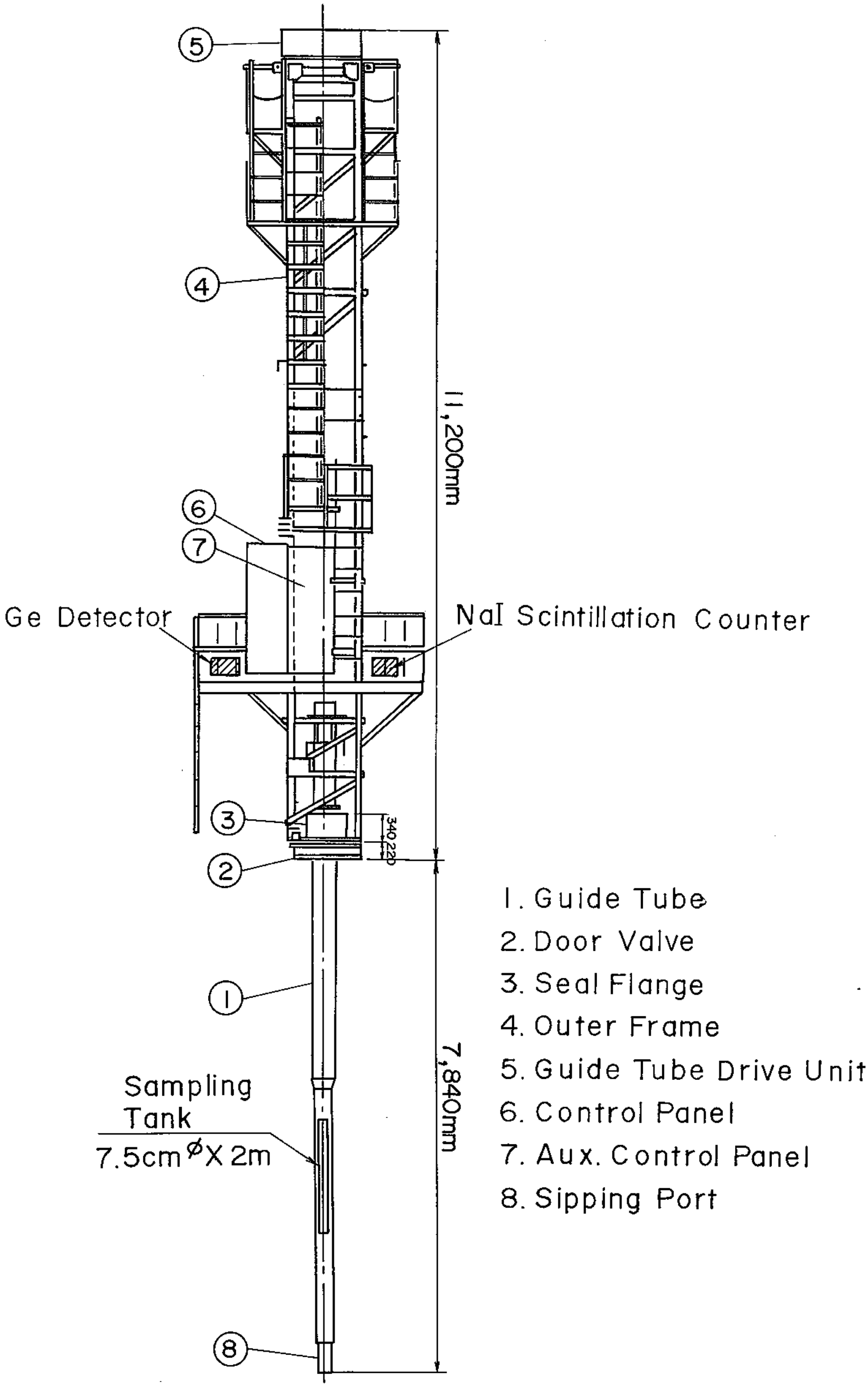


Fig.1.4 Overview of FFDL

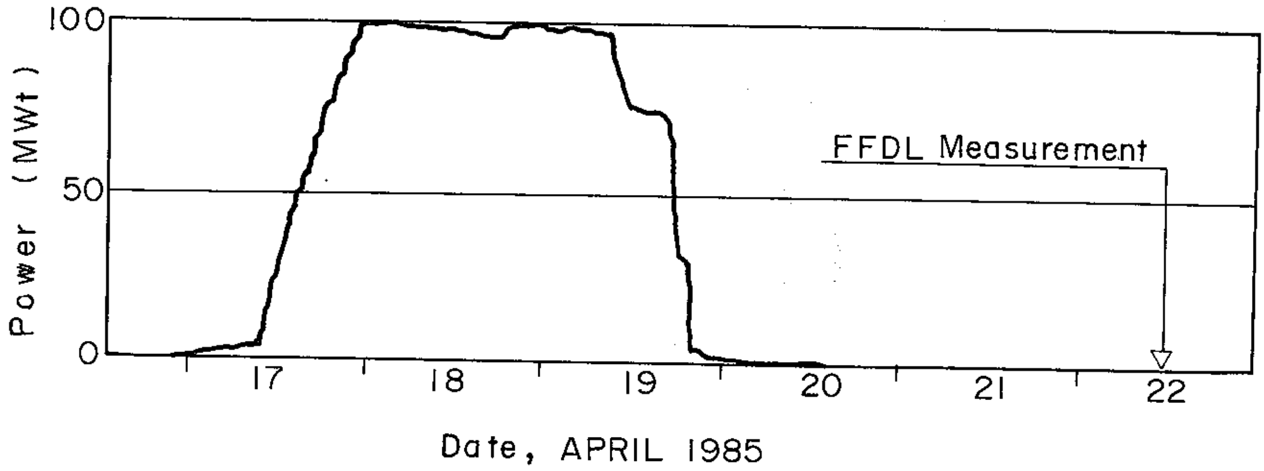


Fig.1.5 Reactor Power History

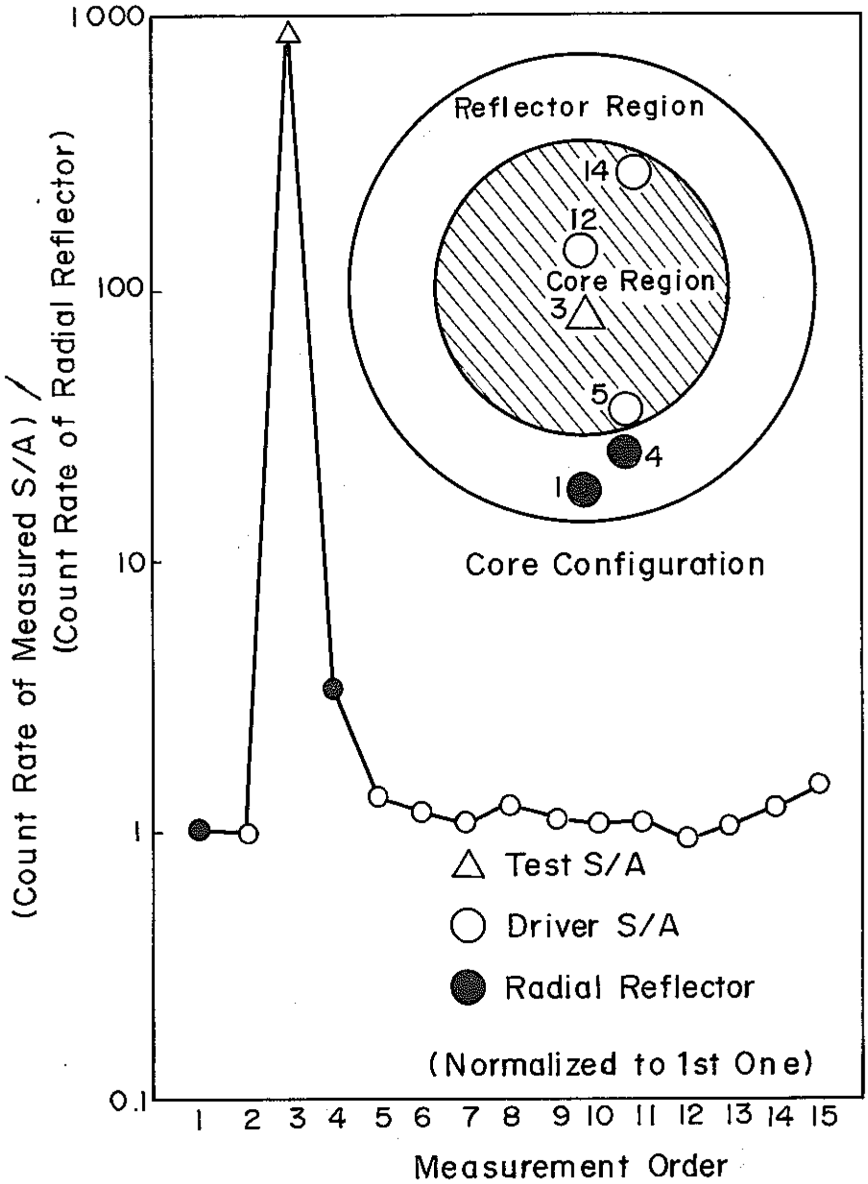
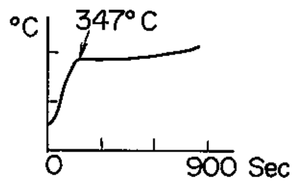
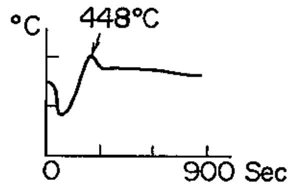


Fig.1.6 Count Rate Ratio of ^{133}Xe γ-ray (81 keV Photo Peak) from Examined Subassemblies

Tabel.2 JOYO Natural Circulation Test (MK-II core)

Test		Test Conditions			Test Results		
		Power	Main Pumps		Central Subass. Outlet Temp.	Loop Flows	
			Pri.	Sec.		Pri.	Sec.
Steady State Test	TEST-II A	1 MW	15% Flow ↓ Stop	40% Flow ↓ Stop		~1 %	1~3%
Transient Test	TEST-II B	30MW ↓ Scram	100% Flow ↓ Stop	100% Flow ↓ Stop		~1.5%	~3 %
Steady State Test	TEST-II C	2 MW	15% Flow ↓ Stop	40% Flow ↓ Stop			
Transient Test	TEST-II D	75MW ↓ Scram	100% Flow ↓ Stop	100% Flow ↓ Stop			
	TEST-II E	100MW ↓ Scram					

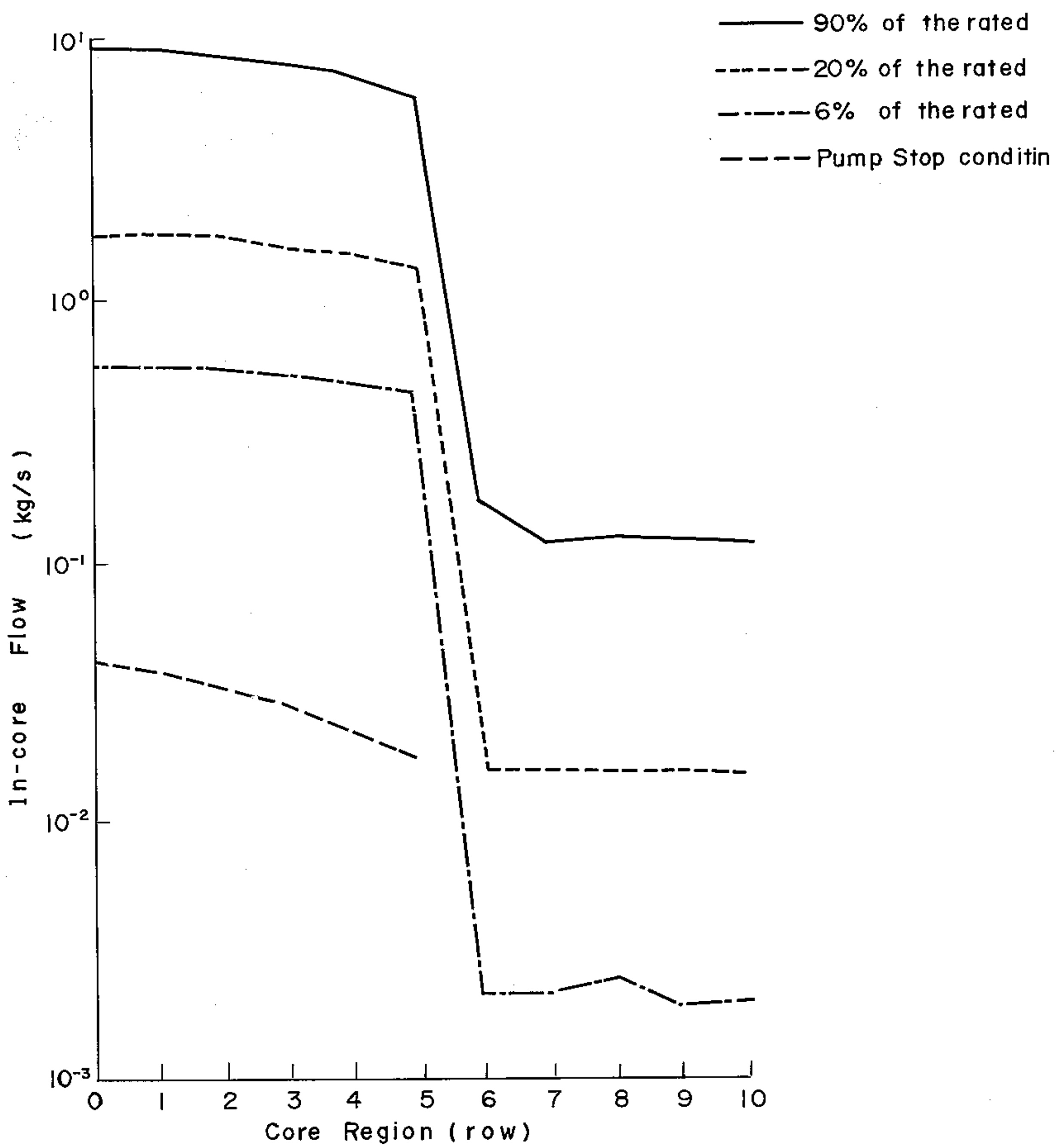


Fig.I.7 In-core FlowRate Distribution

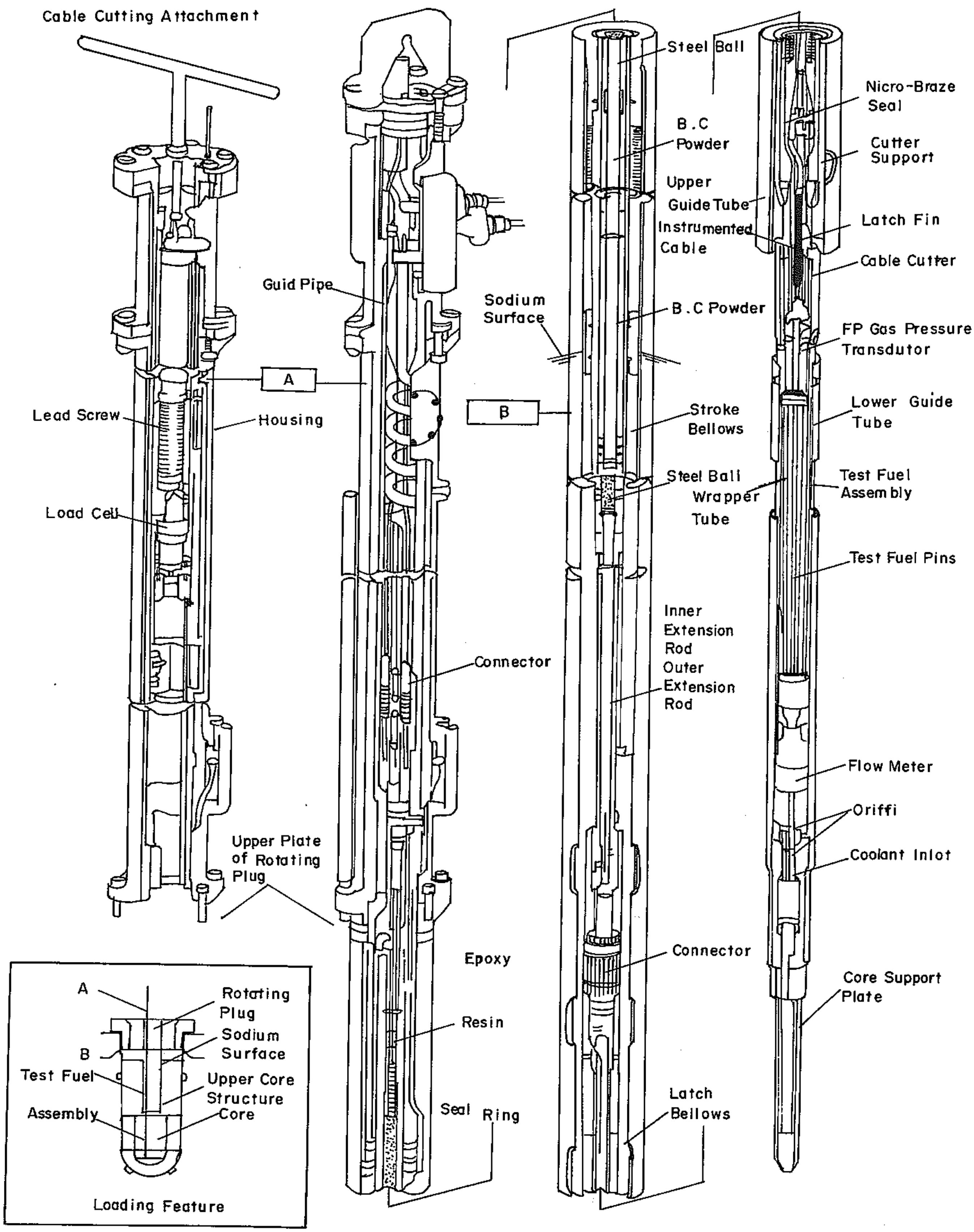


Fig. I.8 Cutawag Views of the INTA

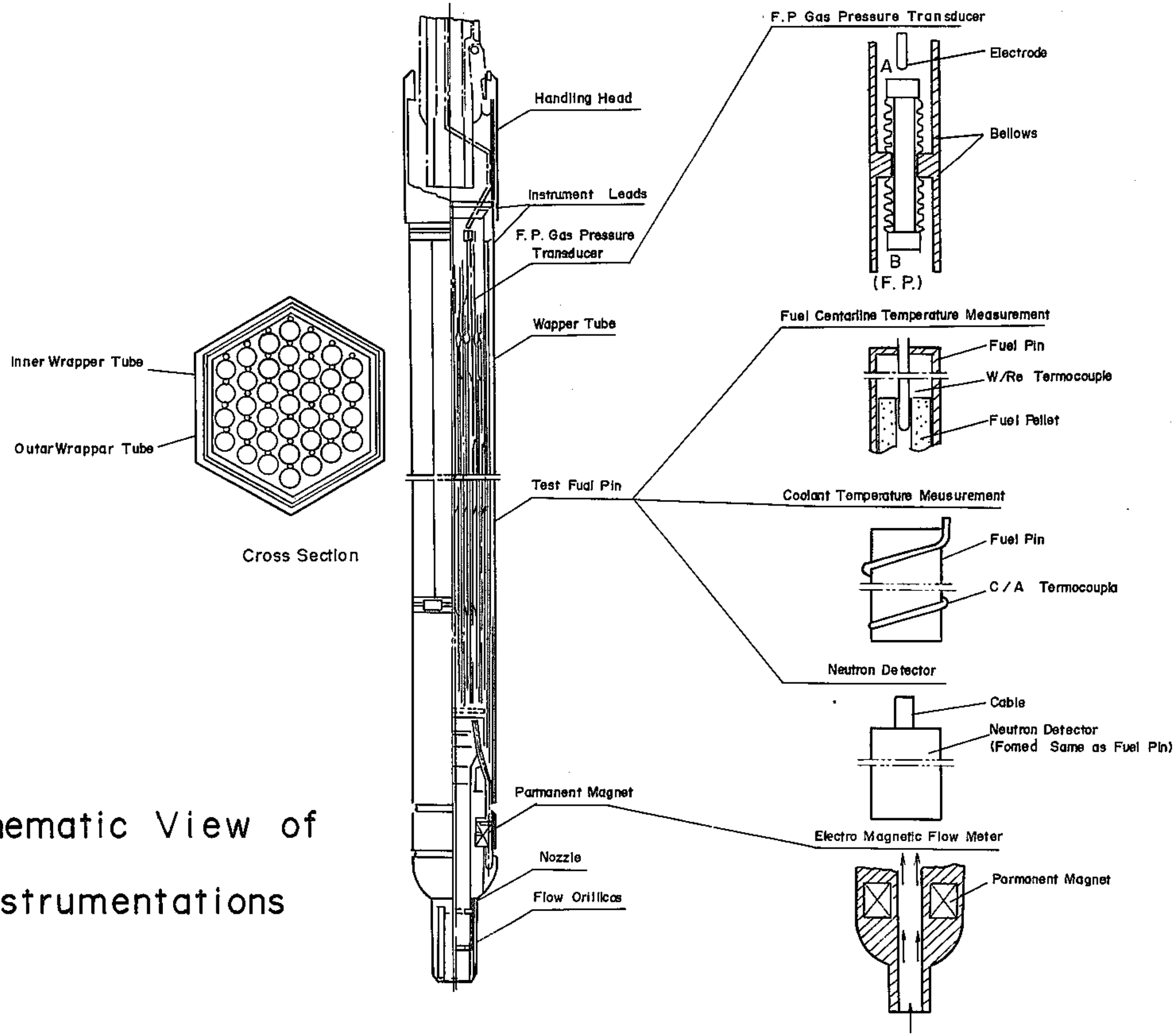
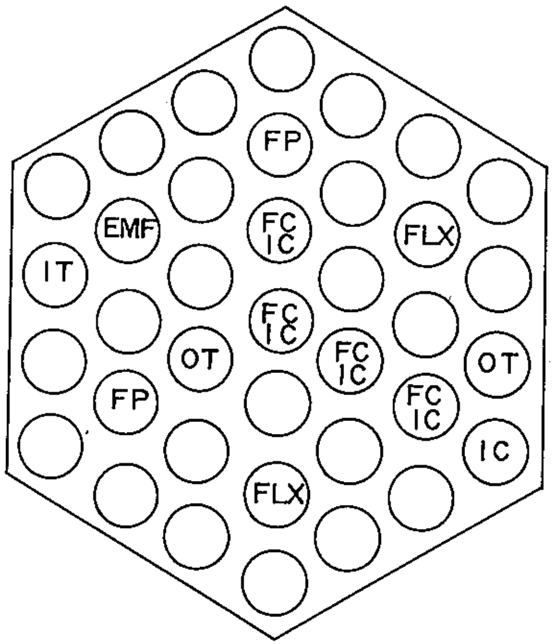


Fig. 1.9 Schematic View of INTA Instrumentations



FP	: FP gas pressure transducer	(2)
FC	: Fuel centerline thermocouple	(4)
IC	: Spacer thermocouple	(5)
IT	: Inlet temperature thermocouple	(1)
OT	: Outlet temperature thermocouple	(2)
FLX	: Flax monitor (Dummy rod)	(2)
EMF	: Conduit for the EMF leads (Dummy rod)	(1)

Fig.I.10 Arrangement of the Sensors Attached to the Fuel Rods

2. Prototype Fast Breeder Reactor Monju

2.1 Summary

Fast breeder reactor Monju plant is a nuclear power plant of medium size, which is being built at Shiraki in Tsuruga City, Fukui Prefecture, approximately 400Km west of Tokyo. Monju is a prototype reactor, which stands between the experimental reactor and the demonstration reactor in the development program of the fast breeder reactor, and aims to attain technological advancement and economic prospect towards the establishment of commercial viability of future nuclear power plants.

Power Reactor and Nuclear Fuel Development Corporation (PNC) undertook the design of Monju and the selection of its site in 1968, and in 1970 Shiraki area was chosen as a prospective site. Since them PNC conducted an environmental assessment and safety evaluation. The environmental assessment was completed by the local authorities of Fukui Prefecture in June, 1982, and by the government in July of the year. The first stage of the safety examination was made by the reactor regulatory section of the Science and Technology Agency (STA) and the second by the Nuclear Safety Commission, after which approval was granted for the reactor establishement in May, 1983. Preparatory construction work began early in 1983 such as preparation of the building site and the roads around the site.

In January, 1984 PNC contracted with four of the main component manufacturers (Toshiba, Hitachi, Fuji Electric and Mitsubishi Heavy Industry) for design, fabrication, and installation of a containment vessel, design and fabrication of a reactor vessel, and design of the primary and secondary heat transport system, steam generator, electrical system, fuel handling system, etc.,as the first stage.

At present PNC is finalizing the detailed preparation for the licensing approval of the designs and construction methods. Reviews on the design and methods of construction by STA and on the construction plan by Ministry of International Trade and Industry (MITI) were initiated at almost the same time in December 1984 and the first permit was granted from STA in August 1985 and from MITI in September 1985, respectively. The main construction work on site was initiated in Octorber 1985 and now in progress.

3. Demonstration Fast Breeder Reactor DFBR

3.1 Overview

Utilities have been carrying out design rationalization study both on loop and tank type plants. This design study has been conducted under the initiative of FBR Project Office of the Federation of Electric Power companies, the role of which was recently succeeded to the Japan Atomic Power Company. (JAPC)

Power Reactor and Nuclear Fuel Development Corporation (PNC) is playing a role of consultation and giving suggestions to the utilities, design study, especially in the area of matters concerning with core design, shielding design, high temperature structural design and safety design. PNC also started the large scale reactor design study to develop the key technologies.

3.2 Design Study of DFBR

The utilities had carried out the conceptual design studies under cooperation of ten private electric power companies until 1983. The study consisted of three phases. Regarding the study of the loop-type reactor, key concepts of the design were selected in the phase I and the design was reviewed mainly from the standpoint of operation and maintenance in the phase Ⅱ. Design specifications were established on the basis of the further design of the total system and components in the phase Ⅲ from FY 1981 through 1983. Regarding the study of the pool-type reactor, a preliminary concept definition was carried out studying the design of preceeding plants in phase I and it was reviewed mainly from standpoint of seismic characteristics in phaseⅡ . The key subsystems of the pool-type reactor were designed in phase Ⅲ in parallel with model tests studied by the Central Research Institute of Electric Power Industries(CRIEPI).

Presently, utilities are carrying out design rationalization study in order to survey feasibility of construction cost reduction and to prepare data for determineation of reactor specification.

PNC is carrying out feasibility study for scale up of main parts of a plant based on the design, construction and operation experience of "Joyo" and "Monju", aiming at the power generation cost reduction, as well as banking and arrangement of technical data and informations concerning above mentioned activities aiming at their convenient utilization for specification determinati on.

4. Physics

4.1 Analysis of JUPITER Experiments

The US/Japan Cooperative JUPITER-II Experiments have been analysed in Japan using the JENDL-2 library. A multidrawer model was applied to the analysis of ZPPR-13A, in order to take into account the interference effect between fuel and blanket regions adjacent to each other. The multidrawer effetct obtained are as follows.

K_{eff}	: 0.43 % $\Delta k/k$ increase
$^{238}U(n, f)$ distribution	: 4 % increase in fuel compared to blanket region
Na void worth	: small effect (-3 ~5 %), but relatively large space dependence

Gamma dose distribution measurement in ZPPR-13B/4 was analysed from the standpoint of absolute data basis. Relative TLD responses were satisfactorily reproduced by calculation. The results showed the similar radial dependence to that obtained from reaction rate analysis. Absolute C/E values scattered around 0.91 in this calculation.

4.2 FCA Experiments

A series of the critical experiments has been planned in order to investigate fundamental nuclear characteristics of the axially heterogeneous LMFBR core and to examine the reliability of the current data and method. FCA Assembly XII-1 was built as the reference core measuring the axial nuclear characteristics of the axially heterogeneous core in the experimental program. Axial distributions of reaction rate, sample worth and sodium void worth were measured in the experiment. Power flattening was confirmed in the fission rate measuremens. The experimental results were analysed using JENLD-2 cross section library and JAERI's standard calculation system. In the reaction rate and the sample worth, the differences of C/E values were observed between the internal blanket and the core.

4.3 Development of 3D Sn Code "TRITAC"

The TRITAC is a three dimensional neutron transport code based on the discrete ordinate method. It has been developed for the purpose of solving the reactor core eigenvalue problem. Efforts were made to improve the acceleration technique, and the diffusion synthetic acceleration method was applied to the code. The method was applied not only to the inner iteration but also to the outer iteration. The preliminary result of numerical calculation showed good agreement with the experimental value. The developed code yielded a proper solution in the two-dimensional geometry. Its effectiveness in the three-dimensional geometry has been investigated in some test cases by comparing the results with those obtained by the rebalance method. The computing time was reduced by over a factor 3. The applicability of the present method to a large FBR cores is currently under investigation.

4.4 Shielding Analysis of Demonstration FBR Design

Shielding charateristics of the 1000MWe Loop-Type Demonstration Fast Breeder Reactor has been analysed with the same data and methods that were applied to the analyses of "JOYO" and "MONJU". In the design of the plant analysed here, the inlet nozzle position of the primary coolant pipe is selected to be upper, and the reactor vessel is rather large in spite of the loop-type reactor. So, the sodium in the large reactor vessel decreases the neutron flux, and it has been proved that the shielding problem does not occur at the outside of the reactor vessel. On the other hand, this coolant of the DRACS (Direct Reactor Auxiliary Cooling System) is unavoidable in this design. Becuase the margin of the flux level at the intermediate heat exchanger of the main coolant required for the primary coolant system, if the smaller reactor vessle is designed.

4.5 JASPER Program

A joint program on FBR shielding research between USDOE and PNC was started in the autumn of 1985. The joint program is referred to as "JASPER", which stands for the Japanese-American Shielding Program of Experimental Researches. The program consists of reactor shielding measurements using the Tower Shielding Facility (TSF) at the Oak Ridge National Laboratory (ORNL) to investigate the properties and behavior of materials, materials systems and fast reactor components to neutron radiation and other associated effects in fast reactors. The three years program shall include the following series of experiments:

(1) Penetration experiment
(2) Gap streaming experiment
(3) Fission gas plenum experiment
(4) Axial shield experiment
(5) In-vessel fuel storage and transfer experiment
(6) Secondary sodium activation experiments
(7) 3-D radiation transport benchmark experiment

5. Research and Development of Reactor Components

5.1 Reactor Vessel and Internal Structure

5.1.1 Hydraulic Tests of Flow Distribution

Experimental studies of hydraulic characteristics in the Monju reactor vessel have been completed with an integral flow model of 1/2.14 geometric scale using water as a working fluid. Currently, the Monju design is being evaluated using the test data obtained.

5.2 Shield Plug

A temperature distribution test was carried out on a simulated reactor upper shield plug which has a scale of approximately 1/3 in diameter and 1/1 in height. It was operated under elevated temperature using sodium. Test result showed the existence of natural convection of argon cover gas in the annular region around the plug. In order to suppress this effect convection restaining plates were employed. After setting of restraining plates, in-sodium testing of the shield plug was carried out from May 1983 to December 1984, and the shield plug was pulled out from test tank and was observed in February 1985. Analysis and evaluation of test results are being continued.

5.3 Primary Pump

Good hydraulic performances of the primary pump for MONJU were already verified by in-water and in-sodium tests with a full-size prototype.

As the final stage of the pump tests an operation test with lowsodium level in the pump casing and non-seal gas tests for pump shaft were carried out on the sodium pump test loop. As a result of the tests, it was not found any trouble.

5.4 Intermediate Heat Exchanger

Water flow tests of 1/5 sector model and 1/2 scale full model of the MONJU intermediate heat exhanger has been completed. The uniform flow distritutions in both primary and secondary side were established with low pressure losses by the flow control devices devoloped in these tests. Further sodium flow test under transient condition was performed to investigate the stratification characteristics in the rising flow region after inlet nozzle which might affect thermal shock rate to the upper tube sheet section. On the other hand, in-sodium life test was carried out on bellows to be used at the top of the downcommer pipe. Non-destructive inspection method was also developed for the tube-to-tube sheet welding.

5.5 Control Rod Drive Mechanisms

Three kinds of control rod drive mechanisms for MONJU have been tested with full scale mock-up under simulated sodium conditions. They are drive mechanisms of fine control rod (FCRD), coarse control rod (CCRD) and back-up shut down rod (BCRD). In-sodium test on the final BCRD model was finished in December 1983, and a seismic test in water was be carried out in spring of 1985. The final models of FCRD and CCRD were manufactured in March 1985 and in-water and in-sodium tests were started in Feb. 1986.

A dynamic behavior and a fatigue tests on shaft seal bellows have been carried out to establish the design basis.

5.6 Fefueling and Fuel Storage System

Refueling system of MONJU consists of in-vessel fuel handling machine(FHM), and ex-vessel transfer machine (EVTM). Testing of prototype FHM in sodium and characteristic test of the shaft seal were completed. Testing of prototype EVTM in sodium has been performed since 1981. In-sodium testing of prototype EVTM was completed in December, 1984.

5.7 In-Service Inspection Equipment

An effort is being made to develop In-Service Inspection Equipment for reactor vessel, its inlet pipes and PHTS (Primary Heat Transport System). Basic examination technique is "Visual". Remote inspection technique with optical fiber scope for reactor vessel are now being developed. As step-II test, γ-ray irradiation tests of fiber scope were completed at R.T to 250℃.

Another effort is being directed to develop ultrasonic transducers for high temperature use on reactor vessel and primary piping system as one of volumetric examination techique.

5.8 Expansion joint for piping system

Feasibility studies of expansion joints for piping system has been started since 1985. In sodium tests of the expansion joints will be started on sodium pump test loop from Feb. 1986. Those expansion joints are horizontal connection and vertical connection types. They are designed for primary system of demo-plant FBR, and size is 42 inches in diameter.

6. Steam Generator System

6.1 Sodium-Water Reaction Study

6.1.1 Leak Hole Enlargement and Leak Propagation Study

The self-enlargement test on a micro-crack defect is under continuation with the SWAT-4 test rig in an attempt to accumulate data of 2-1/4 Cr-1Mo and stainless steels for various sodium temperatures and leak rate. In 1985, tests with the 1.5 mm slit length of the 2-1/4 Cr-1Mo steel nozzles have also been conducted.

With SWAT-2, one wastage test was conducted in the small water leak region under the Monju superheater conditions.

With SWAT-3, a failure progagation test under the evaporator condition was conducted to prove that the tube-burst due to overheating could be avoided in the target tubes when the water coolant flowing inside of the bursting due to the overheating of target tubes with use of the inside water cooling of tubes.

6.1.2 Computer Code Development

Improvement of the SWACS code is under continuation. A new function was validated to analyze thermal response of fluid and structures with a reasonable conservatism by choosing the proper value of the parameters.

6.1.3 Leak Detector Development

Leak detection as well as leak location technique using an acoustis-type leak detector is under development by processing the sound signals with a micro-computer and a large computer. Data were also obtained on the acoustic back-ground noise by using of the 50 MW SG TF.

7. Sodium Technology

7.1 Material Tests in Sodium

7.1.1 Core Material Tests in Sodium

In-sodium test of the fuel cladding (and duct) materials for "MONJU" has been conducted according to the program of core materials development. Test items of these material test in sodium are;

- Tensile test after exposure to high temperature sodium and thermal aging
- Internal pressure creep rupture test
- Corrosion and mass transfer test

As a result of these tests the evaluation method of sodium environmental effect was prepared for the 20% cold worked modified SUS 316 steel in 1985.

A new R&D program especially for demonstration FBR was prepared and R&D activities will be shifted to the modified austenitic stainless steels and advanced alloys (including high nickel precipitation strengthening alloys, high chromium ferritic steels, and oxide dispersion strengthen ferritic steels).

7.1.2 Tribology Tests in Sodium

For the development and licencing of "MONJU" components, tests on some hard facing materials have been performed to clarify the tribological behaviors of contacting and/or sliding parts. The tribology test includes:

- Self-welding test
- Friction and wear test
- Sodium compatibility test

A new test program, mainly for the development and evaluation of cobalt-free hard facting alloys, was prepared in 1985.

7.2 Flow and Heat Transfer

The third heat transfer test at low Peclet numbers is in progress for the 19-rod bundle having a P/D parameter of 1.8. The heat transfer correlation will be obtained from the three sets of heat transfer data for P/D=1.2, 1.5 and 1.8 in the final evaluation.

Measurements of pressure drop for the 19-pin wire-wrapped bundle have been conducted under isothermal and mixed convection conditions for upward and downward sodium conditions for upward and downward sodium flows focussing on friction factor correlations. The similar test for the 37-pin bundle is in progress to study the bundle size effects.

Measurements of the sodim mist content in argon cover gas space at relatively low temperature coditions have been completed.

A water test facility was developed to study natural circulation thermohydraulics in the Monju EVST. A water test for the 1/10-scale EVST model having a simple geometry have been completed. Following the above test, the 1/3-scale, with 1/6-sector model tests are in progress to demonstrate the feasibility of the ex-vessel decay heat removal system by natural circulation.

7.3 Radioactive Material Behaviour and Control in Sodium

This technology area covers studies to decrease the radiation exposure to plant personel due to the contaminated primary sodium systems by the radioactive corrosion and fission products (CP & FP) introduced into the LMFBR primary systems and therefore to improve the maintenance operation safety.

The objectives of the study in this technology area are as follows;

- To establish a computer code for CP behaviour analysis.
- To develop methods of CP trapping.

(a) Development of computer code for CP behaviour analysis

The analytical model for CP behaviour in the primary systems of LMFBR has been developed by using the results obtained from the out-of-pile studies which consisted of the CP transfer experiments and the examinations of metallurgical effects of sodium exposure on the stainless steel test specimens, the experiences in JOYO and the published data by some foreign laboratoties. The computer code named as "PSYCHE" has been developed to evaluate the CP behaviour in LMFBR systems (JOYO and MONJU) and the radiation dose rate arounmd the primary cooling piping systems. The analytical parameters in this model were obtained by using the data of the out-of-pile experiments and moreover in JOYO and then the evaluation for MONJU were performed. The aim of this task is the establishment of a general useful and higher precision computer code for plant planning.

(b) Development of CP trapping method in sodium

The CP trapping materials have been developed to collect the soluble species of CP in sodium. It was found so far that nickel was the most effective material to trap ^{54}Mn and ^{51}Cr, but not effective for ^{58}Co and ^{60}Co. The characterization test of the trapping materials is now in progress.

7.4 Sodium Chemistry and Sodium Purificatin

Some kind of impurities as oxygen, hydrogen and carbon etc. can be introduced into LMFBR coolant during operation. These impurities influence the structural and core materials of LMFBR plant, so the plant must be operated controling the impurities concentration. For the purpose the following studies have been performed.

- To develop the devices and techniques to remove impurities and keep the concentration at a desirable level and to develop the in-sodium chemical meters and the techniques to monitor if the concentration of imputities be kept at the desirable level.
- To understand the behaviour and influence of impurities and to consider how to cope with the situation.
- To develop the regeneration methods of cold trap.

Sodium impurity measurement by metallic specimen equilibrium methods

The impurity level in sodium can be calculated from analyzed value of the inpurity in the metallic specimen by using the equilibrium partition ratio to the impurity level in sodium. This is the metallic specimen equilibrium mehtod. This method was tested for oxygen and carborn. A satisfactory result was obtained by using the vanadium wire method for oxygen.

The tests for carbon have been carried out by using the metallic foil of SUS304L and Fe-12Mn.

7.5 Sodium Removal and Decontamination

Since 1982, decontamination studies for CP in fast breeder reactor are being made in the laboratories at O-arai Engineering Center.

A series of tests has been carried out to develop a decontamination process of radioactive corrosion products (CP) in FBR by using the sample gained from the sodium loop which was contaminated by CP. The deposition characteristics of CP in the sodium cleaning equipment of fuel assembly in Joyo and the chemical decontamination processes have been studied. The results obtained in the present study have been utilized for developing the decontamination process of this equipment.

8. Development of FBR Instrumentation

8.1 Nuclear Instrumentation

8.1.1 In-Core Fission chamber

The micro fission chamber to provide for the instrumented assembly (INTA) has been testing in the core of JOYO since December 1985.

8.1.2 Ex-vessel BF_3 Propotional Counter

The experiments of the ex-vessel BF_3 propotional counter for Monju have been completed in Japan Research Reactor-4. The experimental results satisfied the design specification under the following test conditions.

Accumulated irradiation dose:

Thermal neutron 1.6×10^7 n/cm²

Gamma ray 3.3×10^9 R

Temperature conditions

During irradiation 30℃

Post irradiation 160℃

(Time duration in the post irradiation experiment 30 days.)

8.2 Failed Fuel Detection and Location

A demonstration test of the tagging gas system for locating failed fuel was carried out in JOYO, as described in Chap. 2 of this report.

For Monju, experiments of the cryogenic adsorption system of the tagging gas system is being carried out to determine the design specification, by changing the design parameters.

8.3 Early Warning System for Fuel Failure

8.3.1 Temperature Measurement

Temperature fluctuation due to flow blockage in the pin bundle has been ayalysed by computer code developed.

Fast response thermocouples have been developed for application to above-core instrumentation.

8.3.2 Flow Measurement

New type eddy-current flow/temperature sensors were developed and tested in a sodium loop.

8.3.3 Other Systems

An acoustic detection system is being developed for purpose of detecting some anomalous sound, in particular the onset of local boiling in the core.

8.4 Process Instrumentation

8.4.1 Sodium Flow Meters for Large Piping

Since the permanent magnet type flowmeter was adopted for the flow measurement of the primary and secondary system of MONJU, flowmeter response and calibration method became a major concern. Some tests related to these items are being planned at the O-arai Engineering Center.

A testing of on-site calibration technique using cross-correlation technique of EM Flowmeter noise signals was carried out.

9. Fuel and Materials

9.1 Fuel Fabrication

The fabrication of "JOYO" MK-II fuel is now being carried out at the PFFF (Plutonium Fuel Fabrication Facility).

The fully automated and remote fuel production line is now being installed in the PFPF (Plutonium Fuel Production Facility) for "MONJU" fuel.

9.2 Fuel Pins

Fuel performance of the predefected (slitted) pins tested in JOYO was confirmed to be without fuel temperature increase due to fuel-sodium interaction.

Fuel temperature prediction of steady state fuel pin performance code CEDAR was successfully demonstrated by an instrumented fuel test assembly (INTA) experiment in JOYO.

Mechanical property degradation mechanisms of fuel cladding due to "fuel adjacency effect" was clarified by an atmosphere controlled post-irradiation testing.

9.3 Core Materials

Out of reactor evaluation of advanced austenitic stainless steels for mechanical properties and FCCI (fuel cladding chemical interaction) screening had been completed. Those candidate materials are being irradiated in FFTF.

As the third generation core materials, development of high strength ferritic steels are under way. Cladding fabrication of 12% Cr ferritic steel had been compelted, and is under evaluation. Development of oxide dispersion strengthened ferritic steel for cladding application was initiated.

9.4 Subassembly

Postirradiation examination and evaluation of "JOYO" MK-II driver fuel subassemblies are being conducted. The fuel subassemnblies exhibited satisfactory performance without detrimental subassembly deformation or without any indications of fuel pin breach.

Out of reactor testing of bundle-to-duct interaction for large assembly was initiated. Development computer tomography techniques was also initiated for investigation of the bundle behavior.

9.5 Irradiation Experiments

1) JOYO MK-II

The "MONJU" fuel pins and subassemblies irradiation are being conducted. Irradiation of instrumented fuel pins for the measurements of fuel temperature, fission gas pressure, etc., started by the INTA.

2) Foreign Reactors

Operational reliability testing in EBR-II made a significant progress in fuel pin overpower capability and RBCB (run beyond cladding breach) operation. The results indicated relatively large design margin of overpower-to-breach level. The breached pin diagnosis by fission-gas analysis made a significant progress in the application.

10. Structural Design and Materials

10.1 Development of Structural Design Methods

10.1.1 Structural Analysis Methods

a. Nonlinear structural analysis program

Extension of the general purpose nonlinear structural analysis program FINAS has been made since 1981, particularly with respect to inelastic analysis options, large deformation and dynamic analysis capabilities and output options. FINAS is currently used by users in PNC and fabricators.

b. Simplified analysis method of tube-sheet-shell structures

Simplified analysis procedures combining axisymmetric and plate models are being developed for tube-sheet-shell structures.

Methods of predicting local stresses and strains in the central region of ligament as well as rim-ligament region are explored for thermal transient loads.

10.1.2 Structural design guides

a. Establishment of MONJU design guides and supplements

The following items were discussed on task force basis to establish the improved MONJU design guides and supplements.

- Material strength standard
- Inelastic analysis methods
- Stress reports
- High temperature sodium valves

b. Establishment of application guidelines

As application guidelines of MONJU design guides, the following items have been established.

- Design methods of welds
- Design methods of tube-sheet-shell structures
- Cell liner bolt

c. Improvement of design post-processor

The design post-processor POST-DS based on the MONJU structural design guide for elevated temperature service has been developed and improved. The analysis results obtained by FINAS are easily incorporated into POST-DS.

10.2 Structural Test and Evaluations

10.2.1 Structural element and component tests

In order to evaluate the adequacy of high temperature design rules and analysis methods and also to confirm the integrity of the actual components, the following structural element and component tests have been or are being performed.

a) Creep fatigue tests of elbows in sodium

Tests and evaluation are completed.

b) Fatigue tests and creep buckling tests of T-joints

Fatigue tests and creep buckling tests are completed.

c) Thermal and creep ratcheting tests of pipes and elbows

Creep rachtting tests on 2¼ Cr-1Mo pipes and elbows are cpmpleted.

d) Creep tests of cylindrical shell with axial temperature gradient

Tests are completed.

e) Creep tests of beam with primary and secondary stresses

Tests are completed.

f) Elevated temperature tests of piping beloows

Creep fatigue tests are underway.

Buckling tests are underway.

h) Thermal transient tests of SG tube-sheet Model

The first test and the second tests are underway

i) Thermal transient tests in TTS

The thermal transient test facility for structures (TTS) was has been constructed, and the trial test on vessel is completed. The first test on a vessel model is completed.

10.2.2 Thermal and hydraulic tests in reactor components

Thermal and hydraulic tests are being performed to capture complex thermal boundary conditions for structural design of reactor components.

a. Thermal stratification tests of 1/6 -scale and 1/10 -scale models of MONJU upper plenum tests are completed.

b. Structural integrity tests of reactor vessel with sodium level

Phase 1 and Phase 2 tests are completed.

c. Thermal striping tests for UCS

Water test are completed

Sodium test are completed

10.3 Structural Material Test

Research and development (R&D) on structural materials tests in air, sodium and irradiation environment has been conductedto refine and/or revise the "MONJU" material strength standard. The test items in air and in sodium environment necessary to the design of fast breeder reactor are shown in Fig. 11-1.

The new R&D program for the tests in an air and a sodium environment is called the "Capella" program, and it is currently being performed. The Step-1 program (1985-1987) of the "Capella" includes both R&Ds for "MONJU" and for future reactors especially demonstration FBR.

The neutron irradiation tests are being conducted according to the neutron irradiation program.

10.3.1 Structural Material Test in Air

In-air structural material tests for the development of the "MONJU" strength standard have been conducted since 1977 as mentioned below.

In 1977 through 1978, basic mechanical properties on typical candidate materials for "MONJU" components were tested to compare the design allowable stresses of ASME Code Case N-47 including welded metals and joints. The following tests were conducted.

- tensile test
- creep test
- relaxation test
- fatigue test
- creep-fatigue test

In 1979 through 1981, the above tests were continued to prepare the material strength srtandard for the "MONJU" structural design guide, and some theoretical methods on material strenth behavior were investigated by the following tests,

- creep damage estimation test with strain-hold
- inelastic strain behavior test
- evaluation test on strength of welded-joint
- notch effect test
- others

The test results were evaluated to verify the validity of the descriptions of Case N-47, and were reflected to development of design guide.

The last phase of the tests were carried out in 1982 through March, 1985, and the objective of the tests was to refine the "MONJU" design guide and standard.

From 1985, the Capella program mentioned above was initiated, and is now being taken place.

The Capella program has two principal purposes. One is to clarify the application limits of the present evaluation methods and to improve the accuracy of these methods. The other is to select new proper structural materials for FBRs.

The improvement of the accuracy of the evaluations and the application of new proper mateirals will contribute to the cost reduction for the construction of demonstration FBR.

For these purposes, the Capella program involves developping the following technologies;

- Improvement of "MONJU" technology for cost-down (creep-fatigue life evaluation, strength of weldment, inelastic constitutive equation, and others)
- Design and fablication of large-scale structures
- Modification of material specification
- Application of fracture mechanics

10.3.2 Structural Material Test in Sodium and Water

a. In-sodium Test

Sodium environmental tests on structural materials for "MONJU" have been performed on Inconel 718 alloy which will be used for the thermal striping resistance of the upper-core structures. Tests on the material for the Bellows joint and on carbon transfer behavior in the secondary sodium cricit of "MONJU" will be also continued.

A new set of sodium environmental effect tests, according to the Capella program, are being carried out on possible candidate alloys for a future demonstration FBR. The candidate alloys being investigated are high chromium ferritic steels, and modified type (low carbon and/or high nitrogen) austenitic stainless steels. Testing items included are corrosion and mass transfer tests, carbon transfer tests, and mechanical strength (tensile, creep, fatigue, creep fatigue) testsin sodium.

The corrosion tests of materials for MONJU steam generator are continued in the contaminated, high temperature sodium with NaOH or Na_2O, supposing the water leak in the steam generator.

b. In-water (steam) Test

It was confirmed that austenitic stainless steels such as type 304, 316 and 321 were unsusceptible to S.C.C. in the wet steam including dissolved oxygen and chloride ion up to 200 ppb. As the next step, corrosion tests were initiated to confirm the integrity against S.C.C. for the plugs of steam generator tubings (after plugging) under the same wet steam conditions. The corrosion tests on possible candidate alloys for future demonstration FBR were also initiated under the same wet steam conditions.

10.3.3 Structural Material Test in Irradiation Environment

a. Surveillance Test for "JOYO" and "MONJU"

Surveillance tests for the primary components of the experimental fast breeder reactor "JOYO" have been conducted to confirm the integrity of reactor by evaluating the irradiation effects of the materials.

Surveillance test data are used to establish the operating program of "JOYO".

The first surveillance tests were completed on the materials of core barrel, core support plate, reactor vessel and safety vessel of "JOYO".

b. Research and Development Tests

Research and development tests have been conducted on the structural materials (such as SUS304) steel for primary components of "MONJU" to evaluate the irradiation effects on the mechanical properties up to the end of design life time and to introduce the irradiation effect rationally to the materials strength standards for "MONJU".

Both forged and rolled SUS304 steels, Inconel 718 were irradiated in "JOYO" with SMIR (Structural Materials Irradiation Rig).

Uniaxial in-pile creep test on rolled SUS304 steel has been conducted in JMTR (Japan Material Test Reactor) to compare its results with those of post-irradiation creep.

Another R&D test for the demonstration FBR has been conducted to make clear the relationship between creep rupture strength and metallurgical variables such as chemical compositions, grain size, production process etc.

c. Miscellaneous

The new facility (Material Monitoring Facility 2: MMF-2) was initiated started to test irradiation materials in April, 1984. This facility is equipped with five uniaxial creep test machines, two creep-fatigue machines andothers for irradiated structural material tests.

10.3.4 Data Banking System

Material test data are compiled by the specified data coding sheets, and Data are input into the computer system for data banking, SMAT.

The SMAT system was developed to meet the requirements of efficient processing of large amounts of data.

The system has the following features:

1) An integrated data bank system capable of checking, creating, and

retrieving all required data.

2) Easily modifiable processing structures made possible by the use of macro instructions to handle routine jobs.

3) The dynamic and detailed search and display functions available through commands.

The SMAT system is being improved by user's requirments.

11. Safety

11.1 Thermohydraulics

11.1.1 Thermohydraulic Analysis Codes

1) Multi-Dimensional In-Vessl Thermohydraulic Analysis

The work using COMMIX-1A code in this period includes : improvement of the numerical diffusion and the turbulence model, addition of the routine to calculate material transport and the DRACS model and a systematic analysis of the thermal stratification experiments in Japan. In-vessl natural circulation analysis was also conducted using the COMMIX-DRACS version.

2) Plant System Transient Analysis

The Loss-of-Piping-Integrity accident of Monju was analyzed for the verification of the improved version of SSC-L code.

The evaluation of postulated Loss-of-Heat-Sink accidents is in progress. The heat loss from piping walls proved to have important effects on such analysis.

3) Fuel Subassembly Analysis

ASFRE code has been developed into the new version, ASFRE-Ⅲ in which the reliability and efficiency of transient calculations are improved and wire spacers are better represented. SPIRAL, a three-dimensional, single-phase distributed parameter code for a wire-wrapped pin bundle is still under development. At present, laminar flow around a single wire-wrapped pin can be analyzed.

For two-phase flow, the numerical method of the SABENA-3D has been improved to reduce calculation time. ARMADA, an extended version of the SABENA code is now under development, which deals with complex multi-channel boiling dynamics.

11.1.2 Decay Heat Boiling Test

The modificatin work of the Sodium Mixing Test Facility to accomodate a DHB test loop has been completed. Preliminary tests were finished and currently boiling tests are underway. A new test bundle (37H) is a 37-ing bundle simulating the full length of Monju fuel subassembly with a chopped-cosine heating profile. Objectives of DHB test are (a) examination of steady examinatin of boiling suppression and/or excursion produced by the interacting buoyancy and two-phase pressure drop effects.

11.1.3 LOPI Test

The construction of new test facility called PLANDTL (Plant Dynamics Test Loop) is planned in the coming two year period, which is capable of providing variety of thermal and flow transient conditions. The first test scheduled to begin in 1987 is a demonstration of the decay heat removal capability under the simulated LOPI conditions of Monju plant. A preliminary design was completed.

11.1.4 Decay Heat Removal Tests by Natural Convection

A 19-pin bundle is used to study natural convectin under such conditions as LOPI and loss-of-heat-sink. The test has just completed in the Sodium Mixing Test Facility. The data have been used to validate codes such as ASFRE, COBRA-Ⅳ, etc. Some of the DHB/37H tests are underway providing similar data.

The decay heat from irradiated fuel subassemblies stored in the ex-vessl storage tank (EVST) may be removed by means of natural convection in the tank in which a cooling coil is inserted. The confirmatory test is planned for this scheme using a simple water model. The test rig is almost completed and the test will begin soon.

11.2 Reactor Safety

11.2.1 Accident Analysis Codes (Whole Core Accident Analysis)

Several computer codes are used and maintained in a study of whole core accidents in Monju and future reactors. These include: SAS3D for initiating phase; SIMMER-II for core disruption phase; and SAME for SAS3D/SIMMER-II data transfer. SAS4A was introduced from Argonne National Laboratory and the verificaton effort is in progress through comparing with SAS3D and analyzing the CABRI and STAR experiments. The newly introduced models in the PAPAS -2S code, i.e.MFCI and transient fission gas behaviors, have been verified through experimental analyses and comparisons with the SAS3D code.

A scoping analysis of the energetic sequence for a future reactor was performed by using the SAS3D and SIMMER-II codes.

Several graphics codes also have been developed for post-processing PAPAS-2S, SAS4A and SIMMER-II.

11.2.2 Fuel Failure Propagation

The PNC-CEA SCARABEE-N Agreement was made between PNC and French CEA in the middle of 1985, and PNC sent a specialist to CEA.

11.2.3 Molten Core Material Interactins

An out-of-pile experimental program on molten fuel jet-material interactions has been initiated to investigate Post-Accident Material Relocation Behaviors in HCDAs.

A test facility to simulate the molten fuel jet by low, melting alloy (up to 500℃) was constructed to obtain the fundamental information on the breakup mechanisms of the molten fuel jet in the coolant and the behaviors of the thermal attack on structural materials. Preliminary tests were initiated.

The large facility to melt simulant materials by induction heating (up to ~2500℃) is also under construction for the future experiments on this program.

11.2.4 Transient In-Pile Tests

1) CABRI

PNC has participated in the joint CABRI project as a junior partner since 1975, stationing a delegate at Cadarache, France.
The tests with a fresh or a low burn-up fuel pin were finished, and those with highly irradiated pins are under way.
Useful information has been accumulated on the behavior of fresh and irradiated fast reactor fuel pins under rapid power transient conditions.
The PAPAS, SAS3D and SAS4A codes have been validated throughthe experimental analyses (pre- and post-test calculations).
The latest activity was devoted to the post-test analyses of LOF-driven-TOP type tests, focusing on the key issues on the initiating-phase energetics : coolant boiling, fuel pin failure, fuel motion and MFCI.

2) STAR

PNC joined the STAR program (Sandia Transient Axial Relocation) at the Sandia National Laboratories. The final shot of the experiment, STAR-7, was conducted in the end of 1985, in line with the PNC proposal. The analyses of the early series of STAR experiment are in progressusing the SAS4A code. The results will be used to improve the ULOF initiating-phase analysis.

11.2.5 Large Scale In-Pile Tests (TRAN)

PNC joined the PNC-NRC study of fuel removal potential during the late initiating and transition phases (TRAN Program at Sandia National Laboratories, Albuquerque).

The final TRAN series experiment, G-2, was performed in November 1985. The test reuslts were used to improve the transition phase analysis through validation of the SIMMER- II code.

11.2.6 PAHR In-Pile Tests

A new model has been developed which can evaluate debris bed behavior quasi-two or three dimensionally. Based on the model, the D&DC series in-pile experiments conducted at Sandia National Laboratories have been analyzed.

11.2.7 Core Expansion tests

An out-of-pile experimental program has been initiated to study the phenomena of core expansion during an HCDA due to a neutronic power burst or disassembly. The first series of the tests is performed by injecting high pressure steam into water through a pin bundle to observe the thermal-hydraulic effects in the upper structure of the core and the upper plenum of the reactor vessel. The test results will be compared with the thermal-hydraulic calculations of SIMMER-II for the code validation.

The construction of test apparatus was completed and preliminary tests were started. The pre-test analyses have been also performed with the SIMMER-II code.

11.3 Sodium Fires and Aerosol Behavior

11.3.1 Aerosol Proof Tests of Components and Instruments

An aerosol proof tests of a pony motor of the primary pump and the prototype mechanical snubbers was completed in February, 1985.

An aerosol proof test of the electrical instruments (i.e. cabel, connector, etc.) and a PAMs (Post Accident Monitors) test started from November, 1985 and will be completed by July, 1986.

11.3.2 Pool Fire Test

A sodium pool fire test using SOLFA-1 (a two-story high concrete cells simulating a reactor auxiliary building) has been completed in air atmosphere. The objective of the test is to study convective and radiative heat transfer from a sodium pool surface to the surrounding walls and to validate the computer code SOFIRE-EMII.

11.3.3 Spray Fire Test

Two spray fire tests ; Run-F1 and Run-E1, under atomospheric condition was completed in September, 1985. Run-F1 was to study hydrogen gas build-up due to a spray sodium and high concentration moisture reaction. Run-E1 was to study sodium combustion for a long period of time (30 minutes) using a 100m³ closed vessel.

11.3.4 Simulation Test of a Leak from Sodium Pipe

The test (Run-E2) was conducted in November, 1985 to simulate a sodium leak from the IHT (intermediate heat transport) pipe.

11.4 Radiological Consequences

11.4.1 Containment Response and Consequence Analysis

The CONTAIN code is developing by PNC in cooperation with Sandia National Laboratories, USA under an agreement between PNC and USNRC.

The SLAM code developed by SNL for analyzing a sodium-concrete reaction was merged into CONTAIN 1.0 to make the total cell model more realistic. The modification for treating water release from a heated concrete was completed. The sodium spray fire module was improved to deal with the motion of sodium droplets in the burning process. A new model of the fission product transport from sodium to the atmosphere was implemented.

The modules for sodium fires, aerosol behavior, water release, and sodium concrete reaction were tested. An application to the Monju plant is under way.

11.5 Probabilistic Risk Assessment

In support of the Monju development effort, a research on probabilistic risk assessment (PRA) is being performed within the scope of a level 1 PRA.

This PRA study began in November 1982 as a joint program of PNC and Energy Incorporated in the United States. In the period of 1985 JFY, phase Ⅳ, which covered detailed analysis on dependent failures, human relaibility, fire, and uncertainty/sensitivity, was finished. In order to utilize for Monju PRA study, PNC is compiling and integrating the operating experiences of the various sodium facilities at OEC, and also concluded a SMA with the US DOE in January 1985 to join the Centralized Reliability Data Organization (CREDO).

Since last year, this study has been gradually extended to model the core damage progression and radio nuclide release from the core in order to evaluate the risk to public health and safety. Such a consequence analysis effort is now under way in the safety Engineering Division of PNC.